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Shielding and Fuel Storage Calculations for GUINEVERE

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SIXTH FRAMEWORK PROGRAMME
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Work Package 2.3: GUINEVERE

Shielding and Fuel Storage Calculations for GUINEVERE

Klaus Noack, Udo Rindelhardt and Anatoly Rogov

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1. Introduction

Within the 6th Framework Programme Euratom the Forschungszentrum Dresden-Rossendorf (FZD) participates in the Integrated Project EUROTRANS, which is partly financed by the European Commission [1]. It carries out various research works in close collaboration with partners of the project cooperation. One of the tasks, which FZD performed last year, is defined in work package 2.3 (GUINEVERE) of the Domain ECATS. Within this work package the low power research reactor VENUS of the Belgian Nuclear Research Centre Mol (SCK-CEN) will be converted into a zero-power sub-critical system, which will be driven by an external D-T neutron source [2]. An accelerator will produce a beam of deuterons, which is led through a vertical channel onto a tritium containing target located in the centre of the core. In the target D-T fusion reactions occur and neutrons with energy of 14 MeV are released. The greatest part of these neutrons enters the core of the system and initializes fission reactions in the uranium fuel. In this way, the source neutrons are multiplied by a factor, which is determined by the degree of sub-criticality of the system. Solid lead imitates the coolant inside of the core and surrounds it as reflector. So, in contrast to the VENUS reactor, the sub-critical system will confine a field of fast and not of thermal neutrons.

The whole GUINEVERE facility consisting of the accelerator, the beam guiding system and the fast reactor core will be constructed in the existing building of the VENUS reactor. Compared with VENUS, the new facility will produce radiation fields in the whole building, which will differ from the former ones. Therefore, the reconstruction process had to be accompanied by appropriate shielding calculations to avoid technical measures, which could be unfavourably from the point of view of radiation conditions. Moreover, the calculation results should serve as basis of the application for licensing GUINEVERE. Shielding calculations were carried out by FZD and independently by Forschungszentrum Karlsruhe (FZK).

Except for the components listed above, a storage for the new fuel elements of GUINEVERE has to be built. This storage must meet the demands on its criticality safety, which are defined by the Belgian Nuclear Safety Authority. FZD made a series of criticality calculations for various versions of the storage on the base of which the final variant can be determined.

The collaboration between both partners was organized in such a way that the entire project was guided by SCK-CEN and FZD carried out the calculations, which were needed to solve several sub-tasks of the whole shielding and storage problem. In the course of the work, calculation results were continuously delivered to SCK-CEN.

This Technical Report shortly describes the calculation models of the main sub-tasks and summarizes the obtained results.

2. Shielding calculations for GUINEVERE

2.1. General features

The general features of the shielding calculations carried out by FZD are the following:

- According to the task defined by SCK-CEN neutron and gamma doses had to be calculated for a hypothetical GUINEVERE-reactor which is operated at a steady state power of 50 W.
- The calculations were done with the MCNP-5 code version 1.40 and using the ENDF/B-VI data library included in the code package, which is distributed by the OECD NEA Data Bank [4]. The data library ZZ ALEPH-LIB-JEFF3.1 (NEA-1745), which was favoured by SCK-CEN, was not used because it does not contain data for gamma production and transport and the negligibility of the gamma doses could not be assumed from the beginning.
- Geometry and material compositions of the reactor and of the building with the existing room structure were defined by SCK-CEN. Moreover, an input file for MCNP containing the geometry and material data of the original building and of the GUINEVERE-reactor has been delivered.
- All changes of the system that were subsequently introduced into the calculation model in course of the work were made by FZD in coordination with SCK-CEN.
- Neutron and gamma doses were estimated within the simulation of a criticality calculation.
- All results represented in this report are normalised to a total intensity of the fission neutron source of the assumed critical GUINEVERE-reactor operated at a power of 50 W. Assuming $\bar{\nu} = 2.51$ and $E_{fis} = 180\ MeV$ [3], this source strength amounts to 4.35×10^{12} n/s.
- In general, the doses were estimated as point values using the F5-tally of MCNP (point-flux-estimator technique). To avoid a bad impact on the estimator statistics the dose points were positioned in a distance of 10 centimetres from a material surface. Only in few cases the F4-tally, estimating the volume averaged flux, was used, in particular, for the comparison with calculation results obtained by FZK.
- The estimated values of point fluxes were transformed to doses by means of flux-to-dose conversion factors. The use of special conversion factors was not prescribed by SCK-CEN. After some tests we used the sets: ANSI/ANS 6.1.1-1977 [4] for neutrons and ICRP-21, 1971 [5] for gammas. The adequacy of the conversion for neutrons was especially tested. To this end, MCNP-calculations were carried out for the benchmark published in Ref. [6] and the results were compared with experimental doses and other calculated values. The neutron doses obtained with the ANSI/ANS conversion factors very well agreed with the measurement, whereas the conversion with ICRP-21 set resulted in an overestimation up to 20 %.
- The heterogeneous core model as delivered by SCK-CEN was transformed into a homogeneous model with quadratic horizontal cross section. The motivation of this measure was the acceleration of the Monte Carlo simulation. Thus, a factor of about three could be gained for a typical shielding calculation. On the other hand, test calculations showed that for the shielding problems, which had to be calculated, this approximation has a negligible impact on the results. This fact will be demonstrated in the next section. The homogenisation was carried out in the easiest way: The nuclei of all material elements, which are present in the core, were homogeneously mixed in the whole core volume without a flux-weighting. As area of the homogenised core the square, which encloses all fuel elements, was

determined. The central channel of the core was excluded from the homogenisation. Fig. 1 illustrates the scheme of the homogenisation.

- As main tool for variance reduction the splitting/roulette technique controlled by mesh-based weight-windows (ww), which were derived by the ww-generator offered by MCNP, was used. Only in few cases exponential transformation was additionally applied. With few exceptions, two iterations of the ww-generator were sufficient to get appropriate weight-windows. Since the theoretically optimal weight-windows vary with the positions of the dose points, those, which are located in certain vicinity, were considered as group of dose points and optimized MCNP-calculations were carried out separately for such groups.
- In spite of the core homogenisation and of the optimisation of the weight-windows by means of the ww-generator very long computation times were necessary to get sufficiently small statistical errors of the doses. Therefore, in the process of the technical development of the project we aimed at a statistical precision of less than 10 per cent. But, for final calculations the results of which should serve as basis for the licensing a precision of ~ 5 per cent should be achieved.

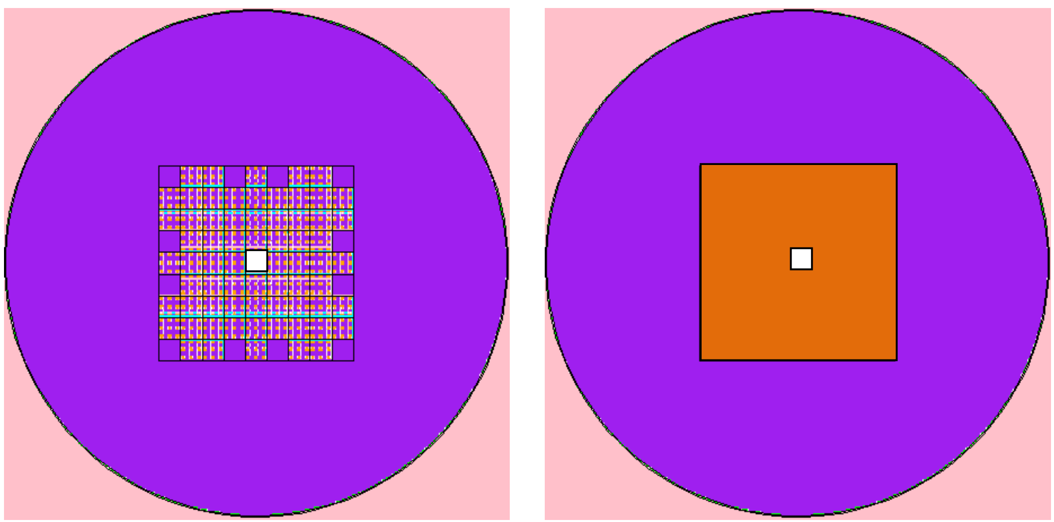

Fig. 1: Homogenisation of the GUINEVERE core; for explanations see text.

2.2 Test calculations

The first sub-task, which has been completely defined by SCK-CEN, is illustrated in Fig. 2. The figure shows a cut-out of a vertical cross section through the room structure around the reactor as modelled by the original MCNP-input data file. As in this figure, we agree on the further use of colours for materials of construction elements as follows: blue - ordinary concrete, magenta - heavy concrete and yellow - polyethylene. The floor between the operator hall and the accelerator hall and the walls in the accelerator hall are new concrete elements, which are planned especially for GUINEVERE. The indicated coordinate system is used throughout this report. Coordinate values will be given in centimetres in accordance with the convention made by MCNP. The z-axis matches the central axis of the reactor. The cut is made as y,z-plane at x=0. The neutron and gamma doses had to be calculated at the four points shown in Fig. 2. Point 1, which is positioned in core mid-plane at x=0, y=-245, z=-70.5, was especially used as test of the influence of the core homogenisation. Table 1 contains the obtained results, where D_n is the neutron dose and D_γ is the gamma dose. The relative statistical error is given in per cent. Due to the thick lead radial and axial reflectors of the GUINEVERE reactor, the gamma dose mainly consists of the secondary component that is generated by inelastic neutron reactions during their transport through the materials.

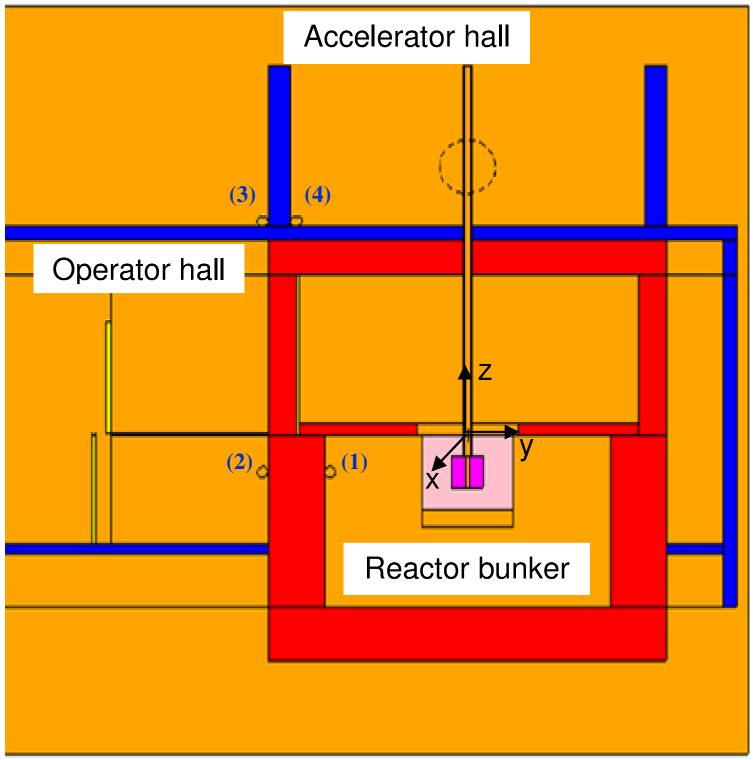

Fig. 2: Cut-out of the vertical cross section through the axis of GUINEVERE.

Tab. 1: Neutron and gamma doses calculated at dose point 1 in Sv/h.

Data library	Heterogeneous core	Homogenized core
ENDF/B-VI	D_n = 2.8 ± 1 %	D_n = 2.6 ± 1 %, D_γ = 2.5x10^{-2} ± 3 %
JEFF3.1	D_n = 3.0 ± 1 %	–

The results illustrate the conclusions which were drawn from the test calculations:

- The neutron doses for heterogeneous and homogeneous core agree within 10 %.
- Both data libraries give neutron doses which agree within 10 %.
- Compared with the neutron dose, the gamma dose is in the range of per cent.

Therefore, in all further calculations the ENDF/B-VI data library and the homogeneous core model were used.

The calculations for **dose point 2** at position (x=0, y=-365, z=-70.5) gave the results (in µSv/h):

$$\mathbf{D_n = 25.4 \pm 2\ \%,\ D_\gamma = 5.9 \pm 7\ \%.}$$

These values illustrate that the secondary gamma dose behind concrete walls generally must not be assumed as small.

2.3. Dose calculations inside of the accelerator hall

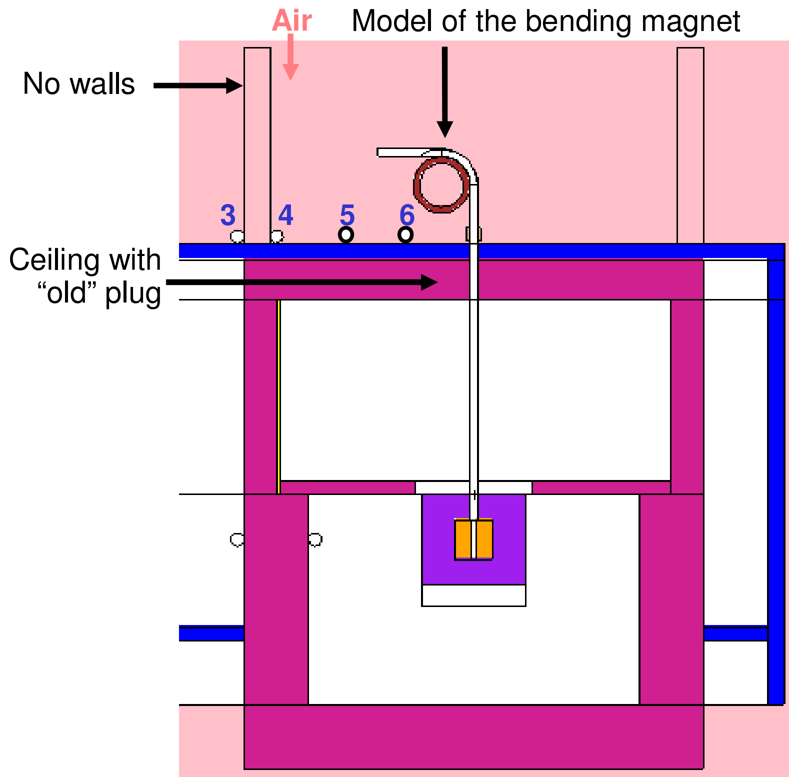

Fig. 3: Cut-out of the model for dose calculations within the accelerator hall.

During the calculations for points 3 and 4 it was decided to remove the walls, which originally have been planned for the accelerator hall. Therefore, the doses were calculated at points 3, 4, 5 and 6 for the modified geometry of the accelerator hall as shown in Fig. 3. They are located at x=0, z=397.1 and y as given in Tab. 2.

In course of the calculations it became clear that the doses at the given points are mainly contributed from neutrons and secondary photons, which are scattered or produced by the beam tube and by the bending magnet. Compared with these dose contributions, those made by neutrons and gammas transmitted through the ceiling turned out to be substantially smaller. Therefore, the modelling of the beam tube and of the bending magnet is important for dose calculations inside the accelerator hall. We developed the models of these components on the base of information and data obtained from SCK-CEN and IPN Orsay. Since the exact modelling of the magnet would demand a very extensive work we, developed a simplified model, which contains the main components as approximated geometrical bodies filled with homogenised material compositions. However, the model represents the given masses of all present isotopes.

Very long computation times were needed to get statistical errors of the calculated doses in the range below 10 %. The gamma doses needed even considerably longer times. The obtained values of the neutron doses are also given in Tab. 2. In parallel to the point dose calculation, for point 4 the F4-tally was used in a sphere with radius of 10 cm. The result was D_n = 20 µSv/h ± 9 %.

Tab. 2: Calculated neutron doses at dose points within the accelerator hall.

Dose point	**3**	**4**	**5**	**6**
y	-365	-304	-200.0	-100.0
D_n (µSv/h)	5.1 ± 8 %	21 ± 8 %	54 ± 9 %	230 ± 6 %

After these calculations it was decided to modify the plug of the beam tube through the concrete ceiling. Therefore, the question about the effects on the radiation doses of this modification was posed. In order to answer it a specific calculation was carried out. The calculation model is shown in Fig. 4. The coordinates of the dose points and the calculated doses are given in Tab. 3.

Tab. 3: Calculated doses at dose points 7 and 8.

Dose point	**Coordinates**	**D_n (µSv/h)**	**D_γ (µSv/h)**
7	x=0, y=-60, z=402	8 ± 11 %	3 ± 12 %
8	x=0, y=-40, z=430	41 ± 10 %	2 ± 11 %

From these results the following conclusions could be drawn:

- The comparison of the neutron doses with those at points 5 and 6 given in Tab. 2 shows that they are considerably reduced in the vicinity of the plug. Obviously, this is the effect of polyethylene and borated polyethylene, which are used in the new plug design.
- The gamma dose keeps substantially lower than the neutron dose, so that the radiation level in the accelerator hall is further determined by the neutron dose. Obviously, the bad gamma-shielding of the polyethylene is partly compensated by the steel components.

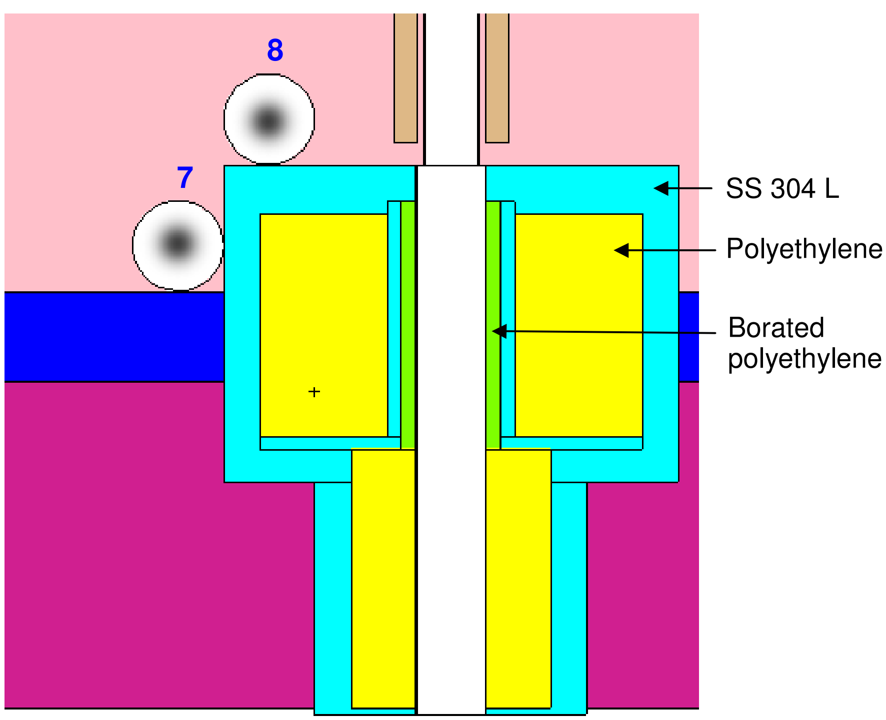

Fig. 4: Vertical cut-out of the calculation model for the new plug of the beam tube.

In order to compare the consequences of different models of the bending magnet in FZK and FZD calculations, we carried out a special dose calculation within the accelerator hall: Analogously to the FZK calculations, with help of the F4-tally the volume averaged neutron doses were estimated for the same detector volumes sitting above the floor of the accelerator room around the plug of the beam tube. Fig. 5 shows the cut-out of the horizontal cross section through the estimation volumes filled with air. The height of the volumes was 20 cm. The x-widths were 170 and 180 and the y-widths 180 and 225 centimetres. The results of both calculations in Sv/h are also given in the figure. The results of FZK are written in boxes. They differ from the FZD results up to a factor 2. The detailed discussion at the Technical Meeting in Brussels [8,9] made clear that the greater portion of this deviation could be explained by a mistake in the FZK model that came from a misinterpretation of design data.

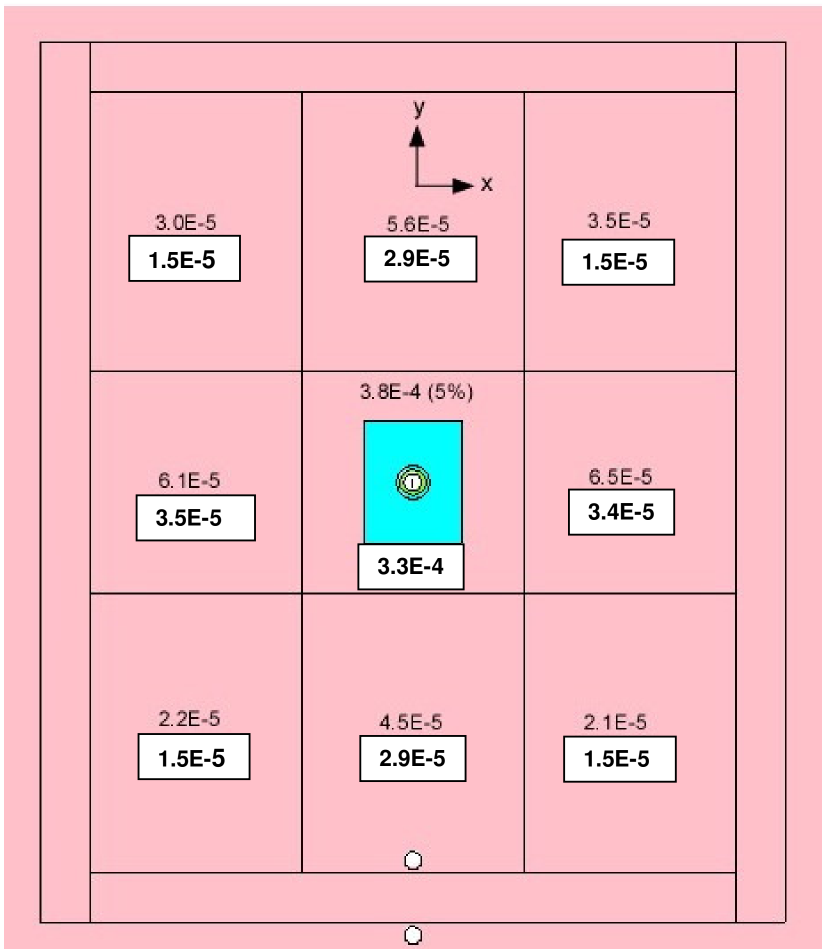

Fig. 5: Cut-out of the horizontal cross section in the accelerator hall showing the subdivision of the estimation volumes used for the F4-tally in a special dose calculation. In addition, the results obtained by FZD and FZK are given. The latter are written in boxes.

2.4. Dose calculations inside of the operator hall

2.4.1. A vertical distribution of the neutron dose

The next series of calculations was directed to the vertical distribution of the neutron dose along the wall of the reactor bunker on the side of the operator hall. The calculation model is shown in Fig. 6. The new dose points 9, 10, 11, 12 and (3) were vertically located above point 2. Table 4 contains their z-coordinates (x=0, y=-365). Point (3) is somewhat higher positioned than point 3 given in Tab. 2.

Tab. 4: Neutron doses calculated at vertically positioned points.

Dose point	**2**	**9**	**10**	**11**	**12**	**(3)**
z	-70.5	50	150	250	300	402
D_n (µSv/h)	25.4 ± 3 %	20 ± 5 %	28 ± 6 %	33 ± 3 %	48 ± 7 %	10 ± 8 %

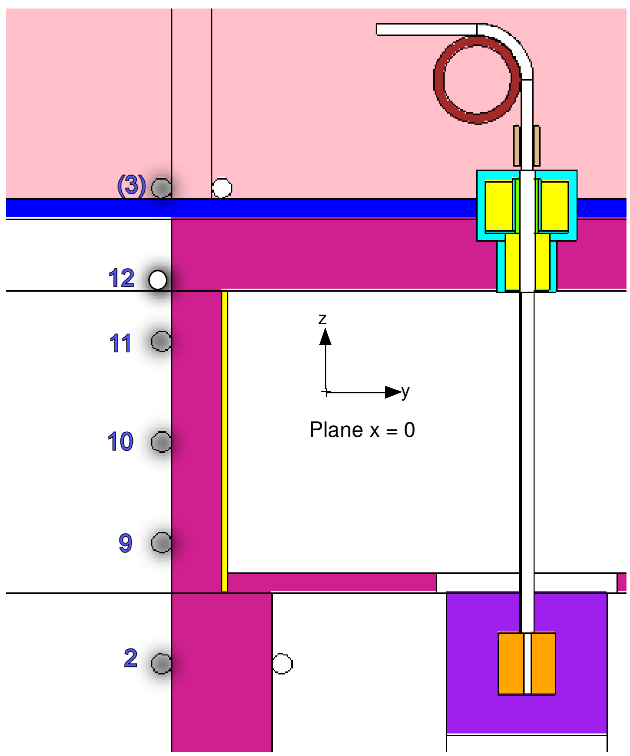

Fig. 6: Cut-out of the vertical cross section showing the positions of the dose points used for the vertical distribution of the neutron dose.

2.4.2. Horizontal neutron and gamma dose distributions

As next step we calculated a horizontal distribution of the neutron dose inside of the operator hall along the wall of the reactor bunker. To this end, along with dose point 2 five further points were located in x-direction at y=-365 in the height of the core centre at z=-70.5. Figure 7 shows the relevant cut-out of the horizontal cross section through the core centre. The coordinates of the dose points together with the calculation results are given in Tab. 5.

Tab. 5: Horizontal neutron dose distribution within the operator hall near to the bunker wall.

Dose point	**2**	**13**	**14**	**15**	**16**	**17**
x	0	-50	-100	-200	-260	-300
D_n (µSv/h)	25.4 ± 2%	38.4 ± 3%	65.3 ± 3%	**84.1 ± 3%**	63.6 ± 3%	52.2 ± 3%

The results showed an unexpected effect: The dose distribution exhibits a maximum between the points 14 and 16. With help of further calculations the cause of this effect could be definitely identified. It turned out that the room above the reactor bunker has a door which is just located in the bottom left corner of Fig. 7, that is, just above the dose points 14 and 15. Unfortunately, the construction of the door allows a neutron streaming into the operator hall from the room above the reactor bunker. The discussion of the observed effect at the Technical Meeting in Brussels [8] made clear that, though the door is in the position “closed”, it does not close well enough for neutrons. The vertical cross section in Fig. 8 illustrates the situation. The “closed” door and the bunker wall form a gap through which neutrons can stream into the operator hall. In particular, the lower gap causes the maximum of the neutron dose observed in the calculation results presented in Tab. 5. In case when a decrease of the radiation level inside of the operator hall turns out to be necessary, then two options should be considered: Either the door will be brought into a position, which better closes the wall for neutrons, or at least the lower gap will be covered by a block of concrete.

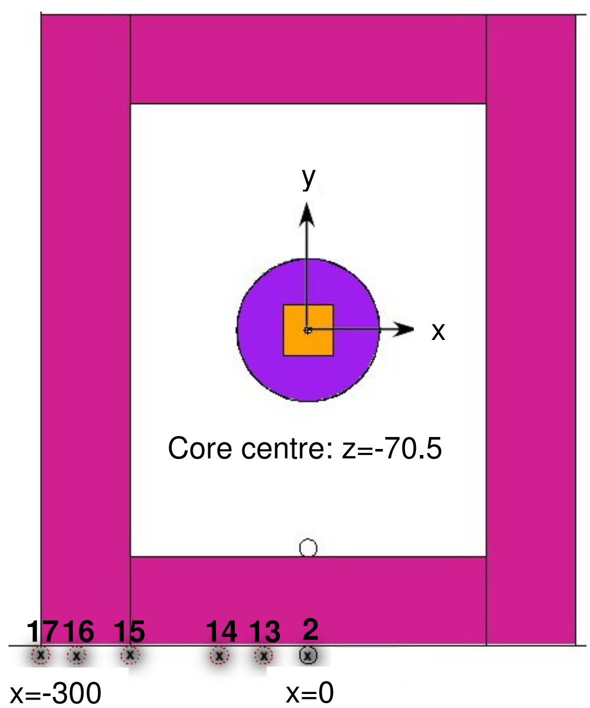

Fig. 7: Cut-out of the horizontal cross section through core mid-plane.

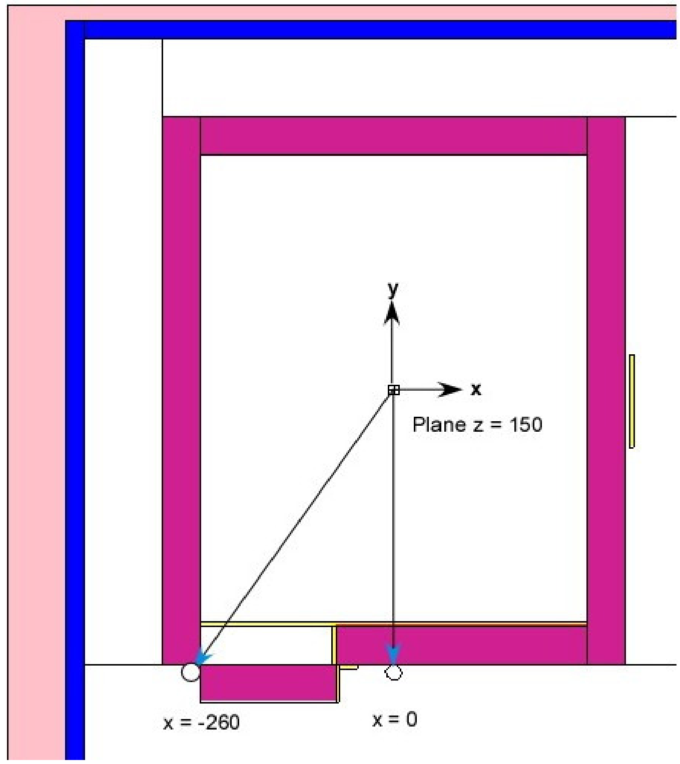

Fig. 8: Cut-out of the horizontal cross section at height z=150.

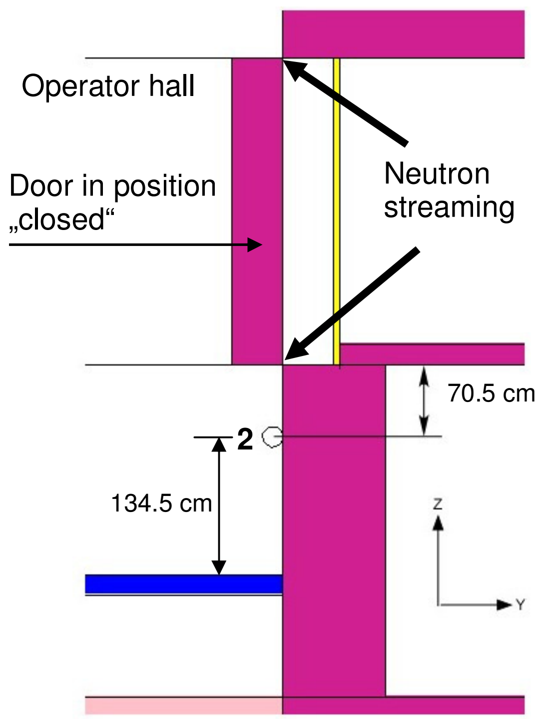

Fig. 9: Cut-out of the vertical cross section at x=-150.

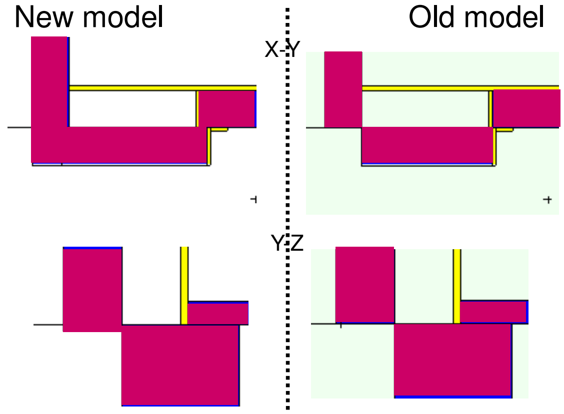

Fig. 10: New and old models of the door in the room above the reactor.

As consequence of the discussion at the Technical Meeting the calculation model of the door was checked and finally modified. Figure 10 compares the old model with the new one, which is more realistic.

With the new door model neutron and gamma doses were calculated once again at points in the same horizontal line near to the bunker wall which is indicated in Fig. 7. The dose points and the results are given in Tab. 6. Compared to the previous calculations, the dose points given in parentheses were newly defined. For better comparison, the results from Tab. 5 were inserted in the table.

Tab. 6: Horizontal dose distributions within the operator hall near to the bunker wall for old and new door models.

		Old model	**New model**	
Dose point	**x**	**D_n (µSv/h)**	**D_n (µSv/h)**	**D_γ (µSv/h)**
2	0	25.4 ± 2 %	7.9 ± 3 %	4.7 ± 10 %
13	-50	38.4 ± 3 %	15.4 ± 2 %	12.0 ± 6 %
14	-100	65.3 ± 3 %	32.3 ± 2 %	25.7 ± 4 %
(14)	-150		41.2 ± 2 %	31.4 ± 5 %
15	-200	84.1 ± 3 %	37.9 ± 2 %	31.7 ± 11 %
(15)	-250		24.6 ± 2 %	16.6 ± 6 %
16	-260	63.6 ± 3 %		
17	-300	52.2 ± 3 %	12.6 ± 3 %	7.8 ± 10 %

The comparison of the calculation results for old and new door models leads to the following conclusions:

- As with the old model, the dose distribution exhibits a maximum just below the door of the room above the reactor bunker.
- The new door model results in lower neutron doses.
- The gamma doses are almost comparable with the neutron doses.

The neutron and gamma dose distributions calculated for the new door model are represented in Fig. 11.

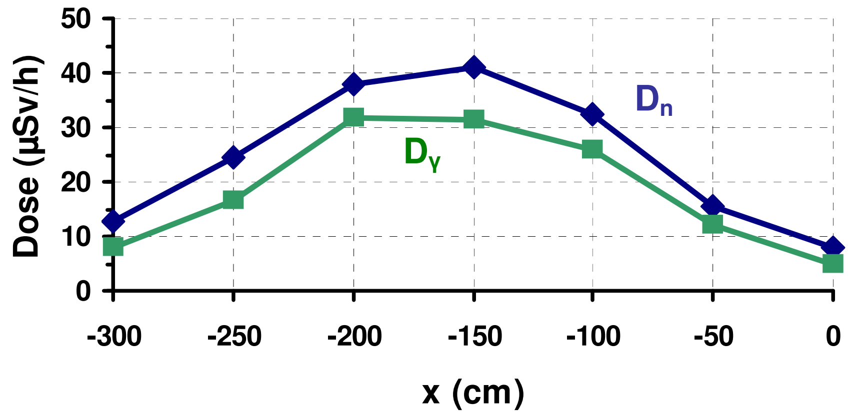

Fig. 11: Neutron and gamma dose distributions at points 2, 13 to 17.

The next series of calculations was made for six points located within the operator hall in the horizontal plane of the core centre. The coordinates of the dose points (red coloured circles) and the calculated neutron and gamma doses with their statistical errors are given in Fig. 12. The points are at a height of 134.5 cm above the floor of the operator hall (see Fig. 9).

It is apparent that in the area beyond the polyethylene wall the doses keep nearly constant and do not noticeably decrease with greater distance from the reactor. The results show that in this part of the hall the total dose should be less than 10 µSv/h with high confidence. The only exception is the neighbourhood of point 18. Here, particularly the neutron dose turns out to be a little raised, obviously, because of the neutron inlet through the door.

3. Shielding and normalisation calculations for the VENUS reactor

3.1. Shielding calculations

SCK-CEN has done neutron dose measurements at the VENUS reactor [10]. Therefore, it was proposed to verify the FZD calculation model by calculating the doses at one measurement point at least and comparing calculation with measurement results. To this end, solely the GUINEVERE fast core was substituted with the VENUS thermal core in the calculation model. The MCNP input file for the VENUS reactor was delivered by SCK-CEN. Fig. 13 shows the horizontal cross-section through the core centre. In the operator hall the dose point 24 was additionally introduced. Dose point 25 is positioned just in front of a PE-shield through which the reactor operators could regularly check the water level of the VENUS reactor. At this point, the neutron dose was measured by means of Bonner Spheres [10].

The scheme of the calculations was the same as for the GUINEVERE shielding, which is described in Section 2.1. The only difference is the normalisation of the MCNP results: For the VENUS reactor a power of 500 W was assumed, i. e. that the total source strength is ten times higher than that of the GUINEVERE reactor, namely equal to 4.35×10^{13} n/s. The calculation results are presented in Fig. 13.

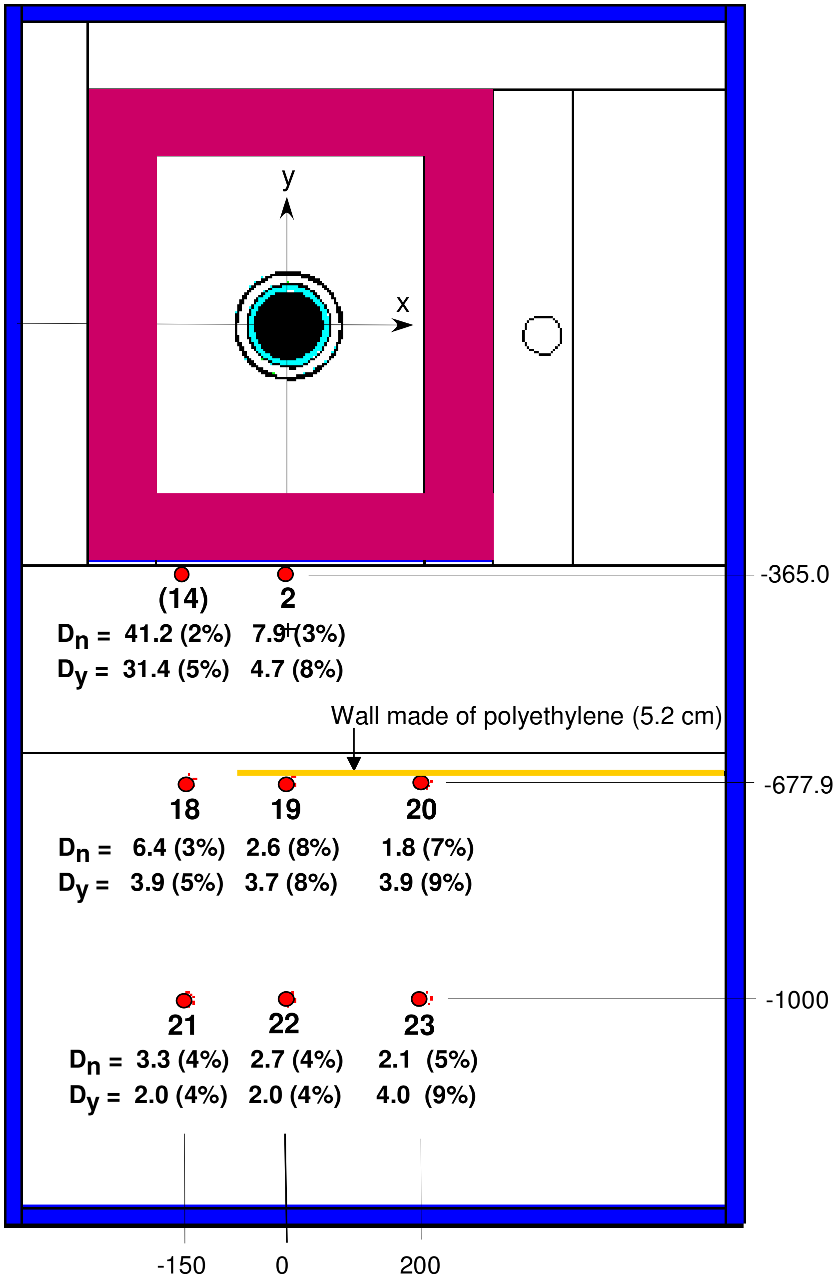

Fig. 12: Neutron and gamma doses calculated at points within the operator hall in the horizontal plane of the core centre.

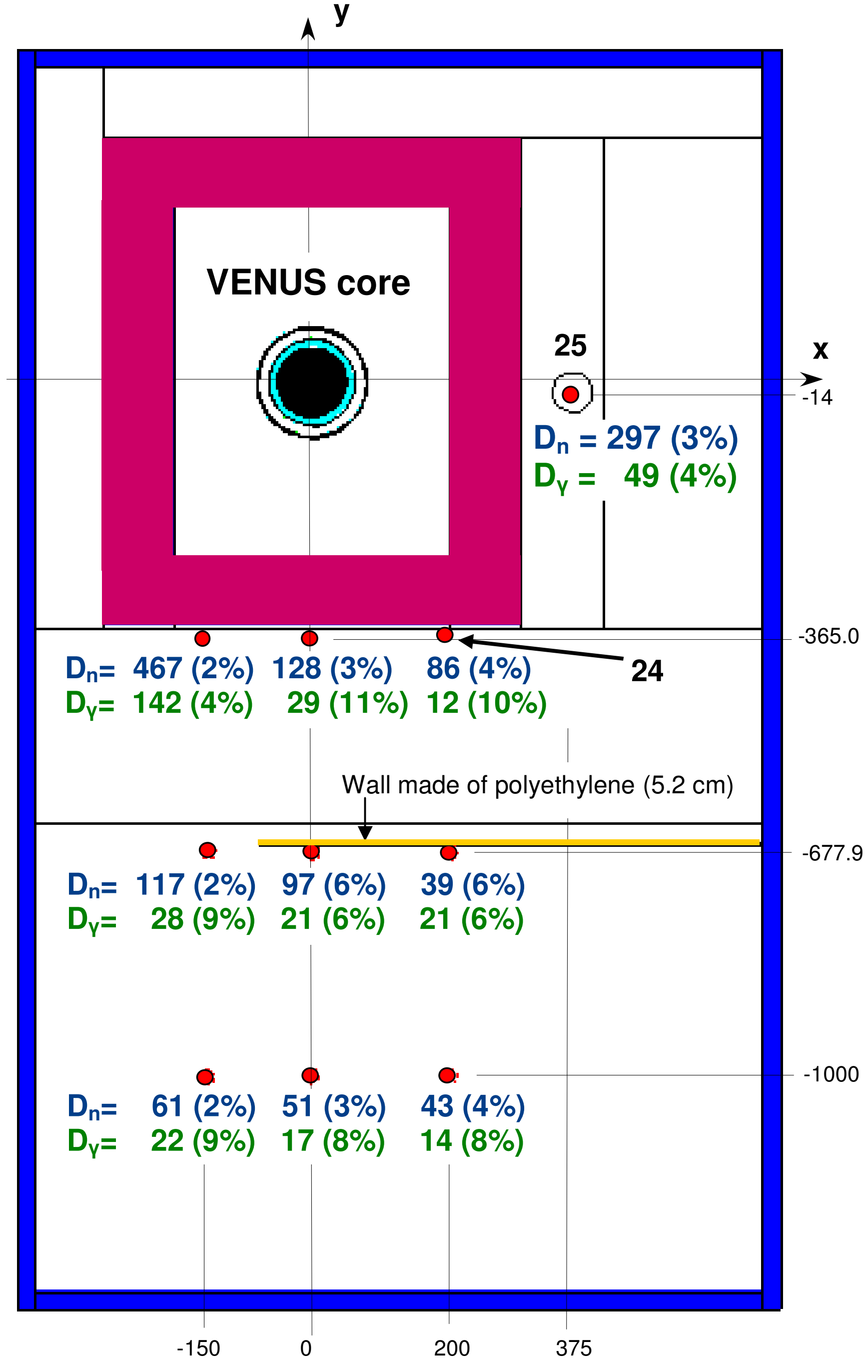

Fig. 13: Horizontal cross section at the height of core centre and calculation results of neutron and gamma doses with their statistical errors for the case of VENUS reactor operated with a fission power of 500 W.

The comparison of the calculation results obtained in both cases leads to the following conclusions.

- In case of VENUS, both the neutron and gamma doses are considerably higher than for GUINEVERE.
- At all dose points the relation between the neutron doses obtained for VENUS and for GUINEVERE is larger than the power scaling factor 10.

– At most dose points the same relation of the gamma doses is less than 10.
– In case of GUINEVERE (50 W), the total dose is clearly less than 10 µSv/h in the area beyond y=-1000.
– In case of VENUS (500 W), both doses are considerably higher than 10 µSv/h.

3.2. Normalisation calculation

The comparison of the measured and calculated neutron dose at point 25 is decisive for the verification of the shielding calculations both for VENUS and for GUINEVERE. The measurements were made at a certain power level of VENUS. On the other hand, the reactor power directly determines the source strength to which the shielding calculations are normalised, see Section 2.1. So, it is necessary to relate the measurement and the calculations to the same reactor power. A low reactor power is very difficult to measure. Therefore, during the dose measurement the counting rate of a fission chamber, which has been inserted in the core centre, was determined. Finally, to get the relation between the results of shielding calculations, which are normalised to 500 W of fission power released inside the VENUS core, and the actual measurement value of the neutron dose the fission rate of the fission chamber had to be calculated. To this end, the configuration of this measurement was modelled as shown in Fig. 14. An empty cylindrical tube made of zirconium was

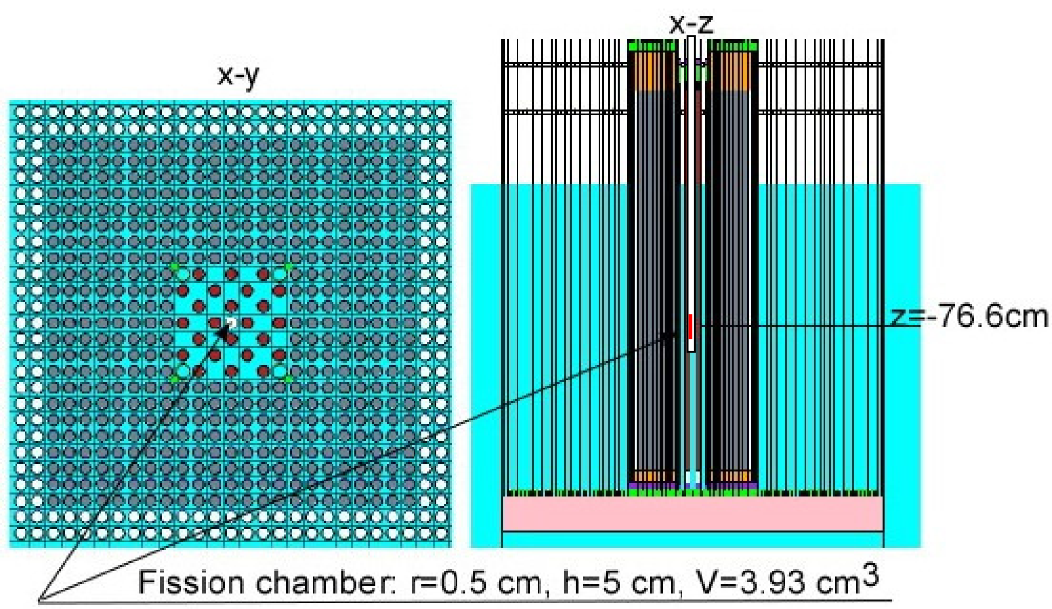

Fig. 14: Horizontal and vertical cross-section through the VENUS core with inserted fission chamber.

inserted at the axis of the core. Its inner and outer radius is 9 mm and 10.1 mm, respectively. Near the bottom of the Zr-tube the fission chamber was placed. It was modelled as a voided cylinder of radius=0.5 cm and height=5 cm. Its volume was taken as estimation volume for the F4-tally of the volume averaged flux. The simulated neutron flux spectrum was folded with the energy-dependent macroscopic cross-section of 400 µg of U-235. Then, the criticality calculation gave the results:

k_{eff} = 1.0006 ± 0.0001,
***fission rate* = 1.70x10^6 fissions/s ± 3 %.**

The calculated fission rate is normalised to 500 W of fission power, which is totally released by the VENUS core.

4. GUINEVERE fuel element storage calculations

4.1. Permanent storage

It is planned to store the GUINEVERE fuel assemblies permanently in the VENUS storage, which is schematically depicted in Fig. 15a. In this concrete box racks loaded with GUINEVERE fuel elements (see Fig. 15b) will be inserted. Each rack can

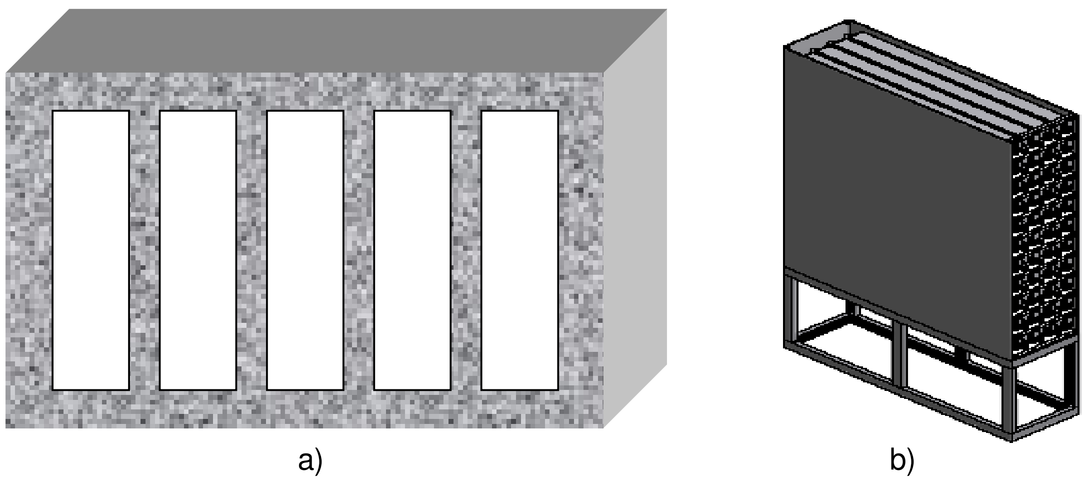

Fig. 15: a) VENUS storage box. b) Rack for fuel elements.

house 44 fuel assemblies, four in eleven shelves. The VENUS storage box is directly at a concrete wall of 20 cm thickness. The fuel assemblies consist of two longitudinal parts: The lower part contains the fuel rods arranged as a rectangular lattice within a lead matrix and the upper part is completely made of lead. The latter forms the upper axial reflector of the core. The assemblies lie horizontally in a rack. The racks are put into the concrete box with the fuel part at the concrete wall and with the reflecting part at the open side.

The racks can be set with shelf heights varying between 12.8 cm and 16.5 cm. With help of criticality calculations it should be found out which shelf height must be determined to guarantee a sufficiently deep sub-criticality of the storage even in a hypothetical emergency case. As maximum value of the effective multiplication factor k_{eff} = 0.90 is given by the Belgian Nuclear Safety Authority. As hypothetical emergency case the flooding of the storage by water had to be considered, however in the special case when the water density has that value, which gives the maximum of the effective multiplication factor. In addition, in accordance with the rules required by the Safety Authority, the VENUS storage had to be modelled as a laterally infinitely long concrete box, which is completely filled with fuel assemblies. Therefore, the MCNP-calculations were carried out within a simulation cell, which is shown in

Fig. 16 for the maximum shelf height of 16.5 cm. The letters L and R both bold printed in parentheses denote the following boundary conditions:

(L) – leakage,

(R) – specularly reflecting boundary.

The latter version realizes the simulation within an infinitely extended row of such simulation cells.

A series of calculations was carried out for the shelf heights 16.5 cm and 14.0 cm with varying water density. Hundred per cent of relative water density represents the case of complete flooding, that is, water is in all empty volumes both inside of the rack and inside of the fuel assemblies. Zero per cent means that air is in all empty volumes. The computed results are given in Tab. 7. They show that in both cases a maximum of the effective multiplication factor appears for diluted water: For 16.5 cm it is at about 50 % and for 14 cm at about 60 % of the normal water density. Both values are already above the given limit. For physical reasons, lower shelf heights must be expected to result in even higher values of k_{eff}. From these results the following conclusions were drawn:

- The shelf height of the racks must be set to its maximum value of 16.5 cm.
- Even in this case, the maximum possible value of k_{eff} is above the given limit and solutions to reduce this value (e.g. the introduction of cadmium foils) have to be considered.

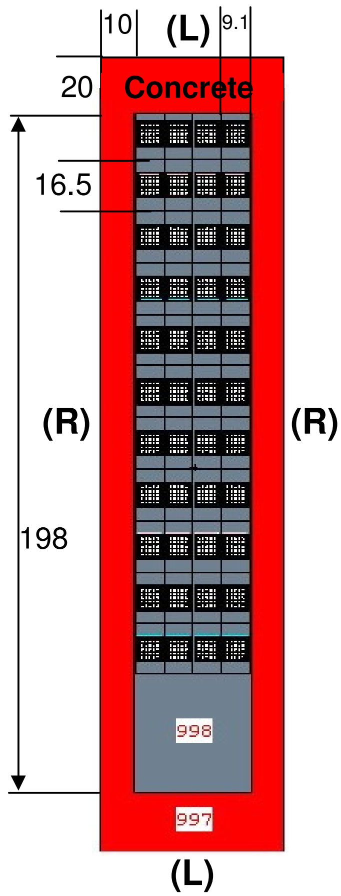

Fig. 16: Vertical cross-section through the simulation cell of the VENUS storage in case of the shelf height of 16.5 cm.

Tab. 7: Effective multiplication factors of the VENUS storage loaded with racks containing GUINEVERE fuel assemblies.

	k_{eff}	
Relative water density (%)	**Shelf height = 16.5**	**Shelf height = 14.0**
0*	0.6298	0.6689
10	0.7504	0.7822
20	0.8437	0.8755
30	0.9016	0.9410
40	0.9273	0.9822
50	0.9306	1.0015
60	0.9172	1.0068
70	0.8955	1.0009
80	0.8722	0.9857
90	0.8447	0.9685
100	0.8215	0.9484

* With air of mass density 1.18x 10^{-3} g/cm^3

The following calculations were carried out only with the shelf height of 16.5 cm. SCK-CEN proposed to introduce cadmium foils of thickness 1 mm horizontally in each shelf and to calculate the multiplication factor for some water densities. A cut-out of the correspondingly modified calculation model is shown in Fig. 17. The Cd-foils were placed directly below the fuel assemblies. The results of the calculations with and without the Cd-foils are given in Tab. 7. They show that the maximum value of k_{eff} is substantially reduced by the introduced cadmium foils.

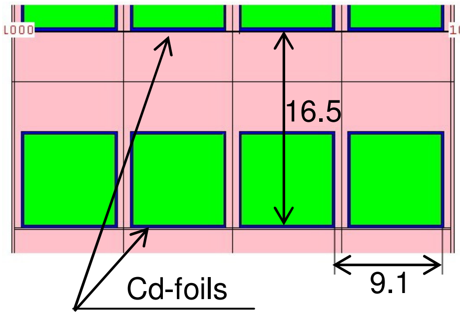

Fig. 17: Cut-out of the vertical cross-section through a rack with fuel assemblies and with Cd-foils.

Tab. 7: Effective multiplication factors for VENUS storage with GUINEVERE fuel elements.

	k_{eff}	
Relative water density (%)	**No Cd**	**With Cd**
0*	0.6298 ± 0.0006	0.5713 ± 0.0005
50	0.9306 ± 0.0006	0.7040 ± 0.0006
100	0.8215 ± 0.0006	0.6282 ± 0.0006

* With air of mass density 1.18×10^{-3} g/cm^3

4.2. Temporary storage

According to the time schedule planned by SCK-CEN for the commissioning of GUINEVERE a temporary storage of the GUINEVERE fuel assemblies is needed. The idea is to put the racks tightly together in a row at a concrete wall with thickness of 20 cm. The horizontal cross-section through the simulation cell is shown in Fig. 18. The calculation results are given in Tab. 8. They show that in comparison with the VENUS storage box the tighter set-up of the loaded racks results in a substantially higher value of the effective multiplication factor. However, the introduction of Cd-foils as considered already in case of the permanent storage in the VENUS box sufficiently lowers the value of k_{eff}.

In spite of this fact, SCK-CEN wanted to consider the effect of additionally introducing Cd-foils vertically between all racks. By this measure, a decoupling of the racks can be expected for the neutrons and, in this way, a lowering of the multiplication factor. In the calculation model the Cd-foils were assumed to be sandwich-like packed with two

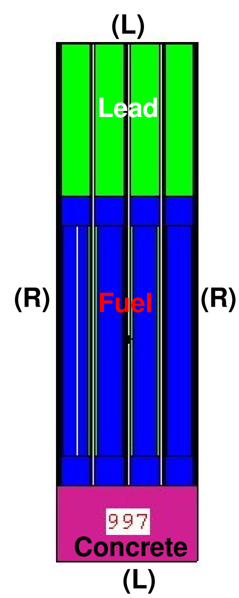

Fig. 18: Horizontal cross-section through the simulation cell for the temporary fuel element storage.

Tab. 8: Effective multiplication factors for temporary GUINEVERE fuel storage.

Relative water density (%)	k_{eff} **No Cd**	k_{eff} **With Cd**
0*	0.814 ± 0.001	0.792 ± 0.001
50	1.145 ± 0.001	0.8406 ± 0.0006
100	0.932 ± 0.001	0.708 ± 0.001

* With air of mass density 1.18x10^{-3} g/cm^3

aluminium foils each with a thickness of 1 mm. Fig. 19 shows cut-outs of a vertical cross-section through the simulation cell and points out the installation of the vertical and horizontal Cd-foils. In the calculation model, the vertical Al/Cd-sandwiches covered the entire area of a rack side.

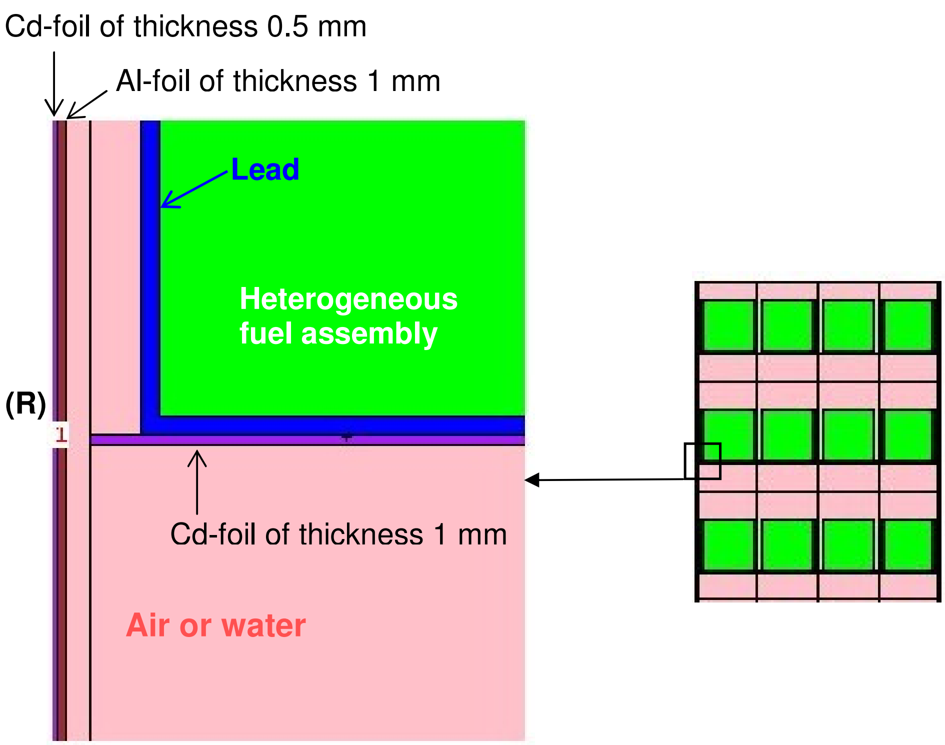

Fig. 19: Cut-outs of a vertical cross-section through the simulation cell of the temporary storage with vertical and horizontal Cd-foils.

The calculation of this configuration for the relative water content of 50 % gave the result:

$$k_{eff} = 0.7960 \pm 0.0006.$$

This means that the installation of Cd-foils between the racks results in a further increase of the sub-criticality. It should be noted, that the sandwiches could be limited to the fuel part of the assemblies only, without a noticeable increase of the effective multiplication factor.

5. Summary

The work done by FZD up to now under the work package 2.3 (GUINEVERE) of Domain ECATS can be summarised as follows:

- A large number of Monte Carlo calculations using the recent MCNP code and data package were carried out in three topical directions: Shielding of the new facility GUINEVERE, shielding of the former facility VENUS and criticality safety of permanent and temporary storing the GUINEVERE fuel assemblies.
- The results of the calculations were continuously delivered to SCK-CEN. They were considered by SCK-CEN when solutions of technical problems had to be decided during the development of the GUINEVERE project.
- In particular, the results of the shielding calculations both for GUINEVERE and for VENUS take an important part with regard to the licensing of the new facility.
- The results of criticality calculations for various versions of the permanent and temporary storage of GUINEVERE fuel assemblies led to solutions, which fulfil the requirements of the Belgian Nuclear Safety Authority and, thus, could be proposed for licensing.

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